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Abstract

STOCHASTIC PROCESSES IN NUCLEAR REACTORS AND MEASUREMENT OF DYNAMIC PERFORMANCE CHARACTERISTICS

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Tabulations are given of the many specific methods of obtaining experimental information about stochastic processes along with data analysis techniques. These range from approximate and inexpensive approaches to very sophisticated and correspondingly precise methods. From this survey it is possible to arrive at certain guiding principles which assist one in selecting the proper method for his needs.

With both theoretical reactor dynamics and the mathematics of stochastic phenomena being highly developed, it is not surprising that in their combination a reliable theory of reactor noise exists. Thus for zero power reactors much success has been had in measuring the neutron lifetime, absolute power level, and shutdown reactivity margin. Data given for the Critical Experiment Station reactor shows that k-effective may be absolutely determined from the ratio of Fourier amplitudes at frequencies f_1 and f_2 of the randomly varying subcritical reactor flux. This is the basis for a simple meter which continually monitors the reactivity of a shutdown reactor as well as calibrates control rods.

In several power reactors it has been found that the source of the random fluctuations in n(t) is traceable to temperature fluctuations in the moderator coolant. In boiling reactors however these are converted to steam void fluctuations which provide the stochastic reactivity function exciting the reactor. Data along with methods of analysis to support these findings are given for the British DMTR, the U. S. Army ML-1 reactor, and the Elk River Reactor.

In the case of boiling reactors many noise spectra have been obtained from the Fourier analyses of the random fluctuations of n(t). These data are summarized and their characteristics tabulated in order to provide a background by which relative stability may be determined on future boiling reactors. Here reactor safety is studied by investigating a resonant frequency in the spectrum of Fourier amplitudes of n(t). It is shown that optimum and preferable analysis procedures exist because of this limited frequency band that is of interest.

While linear point reactor dynamics has been sufficient to characterize virtually all of the stochastic phenomena observed in reactors to date, both non-linear and spatial effects have also been investigated. For the former it has been found possible to generate pseudo-random reactivity excitation signals which measure the desired linear reactor characteristics in the presence of non-linear effects. Since very large power reactors may have regions which are loosely coupled dynamically, stochastic techniques have also been extended in order to investigate these reactors. Data from a zero power reactor having two loosely coupled slabs is given to demonstrate the application these methods have in studying spatial effects.

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Abstract

APPLICATION OF DIGITAL-COMPUTER TECHNIQUES TO REACTOR OPERATIONS

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Over fifty digital-computer-controlled conventional power plants have been started in the U. S. A. The same economic reasons that have led to the selection of computers for control of fossil-fueled power plants should ultimately prevail for nuclear power installations. No completely digitized nuclear control system has yet been installed in the U. S. A., but several plants use digital computers as data-acquisition devices as a preliminary step toward actual control. This paper compares the existing installations, projects the future trends and describes some of the specialized digital control components and techniques at present in use and under development. It includes a philosophic discussion on the role the computer can play in the nuclear power field; safety, reliability and operator-interaction problems of proposed digital control systems are developed.

Studies have been performed by IBM, Daystrom, Ford, Aerojet and others which indicate that the function of the computer can profitably and reliably be extended. Five functions have been projected by these studies as being feasible for nuclear plants. The first function is data logging and alarming. Digital data logging is, or has been, installed at Shippingport, Hanford, General Electric - Nuclear Materials and Propulsion Office, Atomics International, Fermi and Hallam. Computational functions of such variables as xenon poisoning and plant efficiency represent a second simple extension of computer usage once a machine is installed. Big Rock Point is installing a digital computer for fuel-management computations with the claim that this calculation

alone can ultimately pay for the cost of the computer installation. A third function is for the computer to perform diagnostic computations. These calculations are of two kinds: self diagnosis to check that the computer is working properly, and plant-intercomparison types of calculations to check on false scrams or instrument malfunctions.

The fourth computer area is in programming. For the computer to perform its other functions, it must have available most of the detailed information and inputs from the reactor and plant. Once this information is centralized, a logical extension is to use these inputs in programming sequences, such as pre-startup checkout and automatic startup and shutdown procedures.

The ultimate function of a computer installation is full plant control. Control of minor loops such as temperature control is already common in process industries. Adaptive control of the nuclear plant is also feasible because of the large amount of input information available. It is with adaptive control that the plant can always be operated at maximum efficienty, and it is this feature that dictates the economics of computer installation.

Digitized reactor control instrumentation has been developed by Ford and G. E. Some transistorized digital logic systems for safety circuits are currently in use. The NSS Savannah's digital rod drive logic system is compared with digitized motor drives in process work. A few all-digital sensing devices for such parameters as temperature and pressure have been under development and are described.

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Abstract

POISON CONTROL OF THERMAL REACTORS

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This paper covers the use of both burnable poisons and soluble poisons for reactor control.

The burnable poison which has proved most useful to date is boron. Even considering epithermal fission, the ratio of the boron to U-235 cross sections, averaged over all energies of reactor interest, is almost independent of the thermal spectrum. While the helium resulting from the disintegration of boron-10 is objectionable metallurgically, dilute boron loadings appear to be acceptable.

A large number of elements other than boron have been considered for use as burnable poison and appear to have merit for specialized applications. For example, erbium will in some cases result in a more negative temperature coefficient of reactivity, thus enhancing self-protection, and for this reason is employed in the TRIGA reactors.

Very high cross-section elements, such as cadmium and samarium, are sometimes employed as supplementary burnable poisons to increase the shutdown of a new clean core.

Mixing burnable poison homogeneously with the fuel is often inadequate to obtain the objectives of high fuel burnup and long endurance. In this case it becomes necessary to install the burnable poison in lumped or self-shielded form, or to use combinations of homogeneous and lumped poisons.

The application of burnable poisons to the integrated designs of a reactor is stressed, using the Shippingport cores as examples in which the burnable poisons are used to attain extended life and improved power distribution. Practical methods of calculating the effects of lumped as well as homogeneous burnable poisons and design considerations are presented.

The use of combinations of burnable poisons and the use of multi-isotope poisons are discussed. Both local and gross zoning of burnable poisons and fuel are included, as is recent work on shaped burnable poisons.

In the case of uniformly enriched reactors employing fuel management, recent work indicates that by use of suitable combinations of self-shielded poisons a single burnable poison loading can be utilized for the useful life of the fuel, thus avoiding the need to replace the burnable poison when the fuel is reshuffled.

In connection with the application of burnable poisons to practical reactor design, it is of great importance to determine the ratio of burnable poison absorptions to fuel fissions, since this quantity essentially determines the depletion of the poison relative to fuel. In the case of boron, a special problem is presented experimentally in that boron does not activate. The use of various "spiking" materials such as manganese, vanadium and nickel has been explored. Agreement has been obtained in the results obtained from each of these materials. Further, the experimental results are now in good agreement with extensive Monte Carlo calculations.

Neutron absorbing material dissolved as chemicals in the coolant moderator of a water reactor has been used for cold shutdown and during power operation. Reactor core design considerations are noted such as the possibilities for increased power densities, reduced control rod requirements, and implications of moderator temperature coefficient with respect to control and stability of the reactor. The paper covers the means whereby chemical shim is added, controlled, and withdrawn from the reactor.

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Abstract

U. S. EXPERIENCE WITH REACTOR SAFETY AND SITING

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Basically, the only unique potential hazard to the health and safety of the general public presented by the operation of a nuclear reactor arises from the possibility that radioactive materials generated in the reactor in the course of its operation might somehow escape and find their way in dangerous amounts into populated areas. It follows that safety of the public can be assured if means can be found to prevent escape of hazardous amounts of radioactive materials into public areas. Efforts to obtain such assurance are focused on two principal areas:

1. Design, construction and operation of reactors such that they possess adequate safeguards both for prevention of accidents and for minimizing consequences of accidents, and

2. Location of reactors such that isolation distances and other site characteristics are appropriate.

The paper summarizes some of the basic safety principles applicable to design, construction and operation of power reactors and describes the concepts on which our current guides to selection of reactor site rest. Consideration of the siting problem includes a discussion of the roles assigned to "maximum credible accidents" and "exposure dose indices", the possibility of reduced distance requirements as dependability of engineering safeguards improves, a preliminary list of principles relating to the evaluation of individual safeguards, and an indication of the present trend in reactor siting in the United States.

To the present time more than 225 reactors of assorted types and power levels have been built and operated in the United States ... for a surprisingly large total number of accumulated reactor years and megawatt days of operating

experience. For the 20 licensed power reactors alone (four of which are still at low levels of initial operation), more than 1½ million megawatt days of operation have been logged in the six years of time since the first one started. Not a single significant reactor accident (i.e., one that caused consequent damage to people or to the facility) has occurred.

For the other 200 odd assorted reactors: research, test, power (governmental), propulsion and production, a far more impressive operating record (now being compiled for up-to-date totals) has been logged. These reactors have been operated over a span of over 20 years in all sorts of experimental and test programs as well as in steady state condition, and at times when our knowledge was far less extensive than at present. Yet, during this total operation only 15 reactor accidents have occurred which caused either injury to people (three have been killed, all in one accident), damage to the facility, or major unintended increases in power level (without attendant damage).

A brief discussion of these accidents and their implications for reactor safety is given.

As an appendix, this paper contains a description of current administrative and procedural practices in safety evaluation, licensing, inspection and surveillance of licensed reactors in the United States.

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Abstract

CONTAINMENT AND ENGINEERED SAFETY OF NUCLEAR POWER PLANTS

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This paper reviews the safety features utilized in power producing reactors in the United States in operation or under design. Where available, pertinent operating experience is presented for some engineered safety features. In addition, new concepts now being proposed to meet more exacting siting requirements are discussed. This paper does not cover research, experimental, or propulsion reactors.

In essentially all U.S. power reactor designs the fuel is in solid form, usually a high density, high melting point ceramic material encased in a high integrity cladding which resists coolant corrosion and retains all fission products. The coolant system is generally a closed system or one which can be closed to provide a high-integrity closed coolant system. The reactor and the potentially radioactive portions of the primary coolant system are surrounded by massive concrete shielding. In addition, in all cases there is some form of confinement or containment for any material which might escape from the primary coolant system and its shielding.

Fission products can escape into the containment only as a result of some event which provides conditions leading to release of the fission products from the fuel (high temperature) and rupture of the cladding and primary coolant system. In addition, there are possibilities of fuel handling accidents

which are considered separately. Two types of events are generally considered: 1) a nuclear excursion of sufficient magnitude to rupture the cladding and the primary coolant system, and 2) rupture of the primary coolant system leading to loss of coolant and melting of the uncooled reactor core. Either of these types of accidents is considered to be most unlikely. The paper presents considerations regarding these types of accidents as well as fuel handling accidents for various reactors and provides a summary of the inherent features (temperature coefficients, Doppler effects, etc.) and the engineered features (control systems, emergency cooling systems, etc.) utilized in these reactors.

The systems utilized to limit the consequences of the escape of fission products, however unlikely, involve confinement or containment of any material released by various types of sealed cells, steel or concrete containment structures, or low-leakage buildings. There may be two such barriers. Provisions may be made to limit or reduce the volume contained (or the pressure) by means of vapor suppression, pressure relief, sprays, etc. In addition, various types of filtration and spray systems may be utilized to reduce the concentration of radioactive materials which may leak from the containment. Various methods of further reducing containment leakage including pumpback from a secondary barrier may be used. The containment system must also be protected against missiles and must be tested (and retested periodically) to insure leak tightness. These features are presented and discussed for the major types of containment.

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Abstract

MARITIME NUCLEAR PROPULSION EXPERIENCE

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Maritime Administration)

The N. S. SAVANNAH provides a means for the United States to explore the technical, economic, social, and political aspects pertinent to the development of a viable nuclear powered merchant marine. All advantages to be realized from this development are conditional upon complete acceptance of the ship in the ports of the world. This places extreme emphasis on the efficiency and integrity of the reactor containment systems.

A merchant ship must enter crowded ports to pick up and discharge its cargo. Therefore, unlike land-based power reactors, requirements with respect to containment efficiency are greatly amplified. Entrance of the N. S. SAVANNAH into densely populated ports is now made feasible by the containment system, control of fission product inventory, and assurance of ship mobility in the unlikely event of an accident.

Since SAVANNAH's initial startup, a continuing developmental program has been carried on to reduce the early requirement for mobility as part of siting criteria by improving reactor containment. Containment penetrations have been improved, testing techniques have been made more positive and rapid, and pressure suppression systems are being explored.

In addition, the secondary containment exhaust air cleaning system has been augmented with an entrained water separator and better efficiency testing apparatus. The development of more compact, less costly shipboard containment techniques with even higher efficiency is the goal of this work to eliminate mobility as a siting criterion. The development work in these areas and the results obtained to date are discussed.

Other problems associated with the establishment of rules, regulations, and procedures to allow routine reactor operation in the crowded ports of the world are discussed. Reactions by the general populace and authorities are presented.

Before each initial port visitation, the particular foreign government, local port authorities and officials, the U. S. Coast Guard, and the U. S. Atomic Energy Commission review and approve the written analysis and plan prepared for that port. Each visit of the N. S. SAVANNAH is also preceded by a public events team which plans, in close association with local participants, the detailed visitation procedures. Just prior to port entry, this team organizes and supervises all appropriate port services. The development and utilization of these plans in actual port operations are discussed.

Future SAVANNAH plans and the potential impact on future merchant ships are discussed.

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Abstract

SAFETY ASPECTS OF U.S. EXPERIENCE IN SPENT FUEL PROCESSING

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W. S. Frank (General Electric Company - Hanford)

Processing of irradiaced nuclear fuel presents, in addition to usual industrial safety hazards of fire, explosion, and personnel injury by mechanical and chemical means, two unique hazards: the waste products are highly toxic in minute quancities, and the principal product is fissionable. Millions of curies of radioactivity are handled routinely in highly mobile liquid or gaseous states in U. S. fuel reprocessing operations; the overall safety record has been excellent, personnel exposures have been very low, and contamination of the environment has been insignificant. In its overall safety record, the U. S. atomic energy industry ranks with the top U. S. industries; in a typical year, less than one-tenth per cent of USAEC-connected workers received a radiation dose at work exceeding that received in a complete dental X-ray, and none who exceeded this level received a serious dose. In a typical processing operation, the Idaho Chemical Processing Plant handles 1 3/4 million curies of activity per day; less than 2×10^3 per cent of this activity is released to the environment, most of it through the stack. A significant demonstration of containment of high-level particulate activity is to be found in the ICPP Waste Calcining Facility. The gaseous effluent from this plant, processing ICPP aluminum fuel waste since December, 1963, by fluidized bed calcination, has consistently had an activity approximately 0.2 per cent of that permitted by RCG limits.

Several factors contribute to the accomplishment of this record. Facilities are designed to insure adequate radiation protection of personnel and to provide secondary containment of radioactivity in event of difficulties

with process equipment. Processes are designed to provide essentially complete decontamination of all streams before their release to the environment and equipment is designed to provide positive containment of all radioactive streams; critically safe geometry or nuclear poisons are used where feasible. Active safety programs assure effective preplanning of procedures to avoid emergencies, of actions to restrict the magnitude of an emergency, and of actions immediately following an emergency to minimize effects; personnel training assures that employees in processing operations are thoroughly familiar with the plans and procedures; administrative followup assures that the overall program is accomplished effectively.

The overall safety record provides a measure of the effectiveness of these approaches. The few incidents which have occurred were encountered during abnormal or non-routine situations and the effects were confined to the accident area. Five nuclear criticality incidents have occurred in recent years. Two were in a plant processing irradiated fuels and resulted in no facility damage and no significant radiation exposures; three occurred during processing of unirradiated fuel materials and resulted in one fatality and some substantial radiation exposures. Non-nuclear incidents in chemical processing plants have included fires, chemical explosions, and spills of radioactivity in plant areas. Though less widely publicized than the criticality incidents, they indicate the need for continuing emphasis on ordinary chemical process safety. Analyses of these incidents have revealed problems in communication, operator performance, and safety training and have indicated areas in plant equipment where improved operational safety was needed. One result has been the development of more positive mechanisms of administrative safety control in the processing plants. Specific details of the systems vary from plant to plant, but all require (1) a clear definition of responsibilities for safety between the technical and the operations groups, (2) periodic management audits of performance, and (3) competent and thorough review of the safety aspects of proposed new facilities and facility modifications.

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SAFETY ASPECTS OF SPENT FUEL TRANSPORT

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Nelmar J. Rigstad (Phillips Petroleum Co.)
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The transportation of spent fuel, as with other commodities considered dangerous, involves a balance of safety and economy. Of particular importance is protection of the public from radiological hazards in the event of a transportation accident. Information regarding the likelihood and consequences of accidents is so incomplete, however, that justifiable degree of safety is at present a matter of administrative and engineering judgment. Weighing heavily on the side of safety are the public unfamiliarity with the hazards and the absence of physiological warning against some of the hazards.

In the United States during 1957-1962 there were ten reported accidents or malfunctions of casks in transit and in terminals. In six of these there was no escape of radioactive material; in the other four cases, some contaminated water escaped but there was no radiation injury. Fortunately, none of the accidents subjected a cask to severe impact or fire.

The combination of factors involved in designing casks is somewhat unusual, and includes provision for radiation shielding, for removal of heat generated by fission products, for avoiding the escape of radioactive materials in transit, for containment of the contents in a severe accident, and in some cases for prevention of the possible formation of a critical array. Various methods that have been used or considered to accomplish these objectives are discussed.

Methods of calculating resistance of unprotected casks to impact are unreliable; however, drop tests of models and full size casks have shown good correlation, with specimens varying in size from about 30 lb. (13.6 kg) to 30 000 lb. (13 600 kg). The use of shock-absorbing materials to protect casks has been analyzed theoretically, and the conclusions have been verified experimentally.

Work now underway to provide data for calculation of the effects of fire on casks is described. If it is found that providing a high degree of fire resistance is very difficult or costly, careful consideration of the expected frequency and consequences of cask failure will be required. Some accident statistics are presented.

Methods of mounting casks on vehicles may significantly affect the consequences of accident. This subject is under study. Major considerations are outlined.

Certain aspects of design have been covered by the U.S. Interstate Commerce Commission regualtions, and considered by the Bureau of Explosives in granting permits. More detailed performance requirements are set forth in the proposed federal regulation, 10CFR Part 72, which, however, is under review by the USAEC for possible revision. The American Standards Association has published a guide entitled, "American Standard Recommended Practice for Design and Operation of Shipping Containers for Irradiated Solid Fuel from Nuclear Reactors." Important requirements and recommendations of these documents are summarized.

Several procedural devices may be used to help ensure safety of shipment. Among these are routing, type of vehicle, scheduling of time for certain movements, location of a car in a train, speed limitations, and in some cases courier escort. A broad objective, however, is to provide equipment suitable for safe handling by standard methods.

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SAFETY ASPECTS OF THE EBR-II REPROCESSING FACILITY

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The EBR-II Fuel Cycle Facility will be used to study pyrometallurgical methods of reprocessing irradiated fuel and remote methods of fuel refabrication. Substantial safety problems are always posed by the processing of irradiated fuel and the fabrication of fissionable material. Many of the problems are magnified when the material is high-burnup fuel from a high-specific-power reactor and has been cooled only 15 days. On the other hand, the compact unmoderated systems made possible by pyrometallurgy alleviates some problems. The basic safety problems of a facility such as the EBR-II Fuel Cycle Facility can be divided into four classes: shielding and radiation monitoring for personnel protection, radioactivity and contamination control, fire prevention and nuclear safety.

The shielding of the EBR-II Fuel Cycle Facility was designed to reduce the radiation to less than 0.1 milliroentgens/hour in all plant working areas. The in-cell level will be 1 000 000 roentgens/hour. After construction was completed, the entire cell structure was radiographed with a sodium-24 gamma source and with an antimony/beryllium neutron source to check the adequacy of the design. All working areas of the plant are constantly monitored for gamma background and for air-borne activity. All exhaust air from the plant is constantly monitored for gross activity, for particulate activity, for iodine activity, and for sodium-24 activity. Process off-gas is continually monitored for krypton and xenon activity. All monitoring data are automatically recorded and alarms are activated if any abnormal event occurs.

The fission products entering the facility are divided into three streams.

The first, containing krypton, xenon and tritium, is vented to the atmosphere in a controlled manner to insure that maximum permissible limits are not exceeded at any point inside or outside the plant. The second, containing the bulk of the noble-metal fission-products, is deliberately allowed to remain with the fuel (as alloying constituents) and is returned to the reactor. The third, (actually several directly produced solids) containing the iodine, rare earths and the alkaline earth metals, is confined for decay and ultimate disposal (as solids). The only routine aqueous liquid wastes containing fission products are those produced in the analytical laboratories.

Metallic fuels irradiated to 20 000 MWd/ton are in all cases self heating and in many cases pyrophoric. In addition to the fire hazard of such materials, there exists the fire hazard of the metallic sodium used in fuel fabrication and the hazards of the molten magnesium and zinc solvents used in various process operations. These hazards are virtually eliminated by carrying out all operations in an inert atmosphere of high-purity argon gas. The cell gas is over 95% argon and normally contains less that 50 parts/million of water vapor and oxygen, and less than 5% nitrogen. The cell-gas purification system is capable of reducing the oxygen and water vapor in the cell atmosphere to less than 10 parts/million. While the inert-gas atmosphere prevents fires of materials in process, it necessitates special precautions for transfer into and out of the cell so that air contamination is not introduced and so that fires do not occur after pyrophoric materials are transferred into occupied areas.

Nuclear safety - the prevention of inadvertent criticality - is one of the most serious problems of aqueous processes, but it is only a nominal problem in this facility. The safety rules of this facility permit handling of 32 kilograms of 50% enriched fuel per batch; 16 batcher may be in the cell at any one time. This is in excess of an entire reactor loading and approximately seven times the normal plant inventory. All furnaces are designed for radiation cooling rather than water cooling so that not only are aqueous solutions not utilized in processing or fabrication, but water is not present in the cell for any purpose. Similarly, all hydraulic fluids are excluded so that no fluid moderators are present in the cell. Highly flexible storage facilities are provided for in-process storage between stages or while awaiting analytical results.

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Abstract

SAFETY AND SITING CONSIDERATIONS IN THE DESIGN OF THE NFS POWER FUEL REPROCESSING PLANT

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Walton A. Rodger (Nuclear Fuel Services, Inc.)

In establishing the first privately owned spent fuel processing plant--that being built by Nuclear Fuel Services, Inc., at West Valley, Cattaraugus County, New York--a variety of new safety and siting requirements had to be considered. It is the first plant of its kind which is subject to the licensing procedure of the USAEC and thus a number of problems were focused and placed upon the public record for the first time. It was necessary under existing US law to locate this plant either upon state-owned or federally owned land. It was desirable for economic reasons to locate the plant in the north-east portion of the United States near the geographic center of the early developing load of spent fuel. This places the plant in a relatively humid area wherein problems of rainfall, water tables, and water use must be carefully considered.

A site had been selected by the State of New York as an area in which the burial of low level nuclear wastes was to be done. An intensive study of the geographic, meteorologic, geologic, hydrologic, and seismologic features of this site led to the conclusion that it was also a suitable site for the operation of a spent fuel processing plant.

A detailed discussion of the pertinent features of this site is presented. The relationship of the plant, both under normal operation and in consideration of possible operating abnormalities, to its environment is also discussed. The limiting features of plant design and operation in relation to the environment are pointed out.

Some of the special safety problems which were considered in the design of the plant in relation to its siting are: cooling time for decay of short-lived isotopes, the particular problems associated with the use of a mechanical head end, the design of storage tanks compatible with the local site, and criticality requirements. A discussion of each of these aspects is given.

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Abstract

THE U. S. NUCLEAR SAFETY RESEARCH, DEVELOPMENT AND TEST PROGRAM

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The objectives of the United States Atomic Energy Commission Nuclear Safety Research and Development program are to provide a sound technological basis for the rational promotion of nuclear safety in the development, location, design, construction, and operation of reactor systems and associated facilities in land, space, and marine applications. A comprehensive program of analytical, experimental, and engineering scale test investigations is being carried out to develop information in various technological areas. These include: reactor kinetics, fuel-coolant reactions, fission product release, reactor containment, fast reactor safety; engineering field tests of terrestrial and space nuclear systems, post-accident safeguarding and recovery; investigations of radioactive waste treatment, storage, and disposal; and the analysis and evaluation of projects with respect to program objectives. This program is supplemented by, and supports, safety investigations conducted under various specific reactor projects.

A major part of the Nuclear Safety Research and Development program is applicable to the development of civilian power reactors and to the resolution of siting problems in accordance with approved AEC safety policies. Activities related to the increasing use of nuclear devices in space are divided into general research and development, ground tests, and flight tests. They involve actual launches of space vehicles with specially designed payloads and ground tests of space reactors to destruction to directly determine the reactor behavior and consequences resulting from an accident.

The SPERT (Special Power Excursion Reactor Tests) program has produced an extensive background in the transient behavior of light water-cooled and moderated reactor systems including destructive transients in metallic plate-type and UO_2 rod type cores. Fuel meltdown studies in the TREAT facility are providing increased knowledge of the related kinetics and fission product release phenomena. The scope of these studies will be extended with the

proposed construction of the Power Burst Facility which will be capable of testing large fuel assemblies through destruction. A 50 TMW integral engineering test facility is being designed which will permit the conduct of a full-scale loss-of-fluid test in a power reactor. This test, which will be the first of its kind, will be conducted at the National Reactor Test Station in Idaho. Investigations in high-level radioactive waste management have reached the stage at which pilot plants incorporating techniques for the conversion of liquid wastes to solids suitable for permanent disposal have been operated with satisfactory results.

Particular attention has been given in the past two years to the improvement of information feed-back in the planning and execution of the overall program. This has resulted in the establishment of a Nuclear Safety Information Center at Oak Ridge. Mechanisms have been established and used for continuing information exchange with the regulatory and private industrial sectors, and specific industrial firms conducting their own limited investigations of the safety of particular reactor types. Steps are being taken to provide further improvement in the area of information exchange and more intimate participation by industry in the AEC safety program.

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Abstract

REACTOR EXCURSION BEHAVIOR

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Because power excursions in nuclear reactors may, in extreme cases, conceivably result in plant damage and substantial fission-product release, the consideration of excursion behavior is an important aspect of overall safety evaluation of a power-reactor system. In making this evaluation, the inherent physical mechanisms in the reactor that act to quench or enhance the excursion must be identified, their behavior during the excursion established and their effect on excursion behavior understood. These factors, which are in part properties of the type of reactor system, are strongly influenced by specific design features. These must be carefully examined to ensure that undesirable effects are not introduced by design details, to evaluate possible means of initiating accidental excursions, and finally to establish certain specifications or control measures to apply to plant operation or to the design itself.

Much information relevant to the excursion aspect of power-reactor safety is now available from experimental excursion studies conducted under a variety of conditions with a number of different reactor systems, viz: TRIGA, a ZrH moderated system; Treat, which is graphite moderated; Kewb, an aqueous homogeneous system; Godiva, a fast all-metal assembly; Spert I, which is water-moderated with pin or plate type fuel; Spert II, which is heavy-water moderated with plate-type fuel; and Spert III, which is a high-pressure water-moderated system with plate-type fuel. It was demonstrated that various quenching mechanisms were effective in the self-limitation of deliberately induced power excursions; measurements of the magnitude of their reactivity effects were also obtained. These mechanisms were radiolytic gas production, geometric changes in fuel element configuration due to heating, density

changes in coolant-moderator by several heat-transfer mechanisms, changes in scattering cross-sections, Doppler effects and, in several cases, disassembly of the reactor. Most of these mechanisms are important in current power-reactor types; however, the magnitude of the reactivity effect is affected by the reactor type and even by the plant operating conditions (as was shown in experiments with Spert II and Spert III). As expected for these systems, increasing the initial temperatures and pressures in a test series resulted in significant, but not extreme, increase in the energy release. High coolant-flow rates also produced increases in energy release but not in fuel temperature. Most importantly, simulating power-reactor conditions did not significantly decrease the capability of these plants to safely self-limit excutsions induced by reactivity insertions considerably in excess of prompt critical. The results with all systems show remarkable similarity in the way in which burst shape, peak power and energy vary with inserted reactivity and have remarkably simple mathematical descriptions. These features are not expected to be qualitatively different in power reactors. It is particularly important that, in the TRIGA, Treat and Spert I oxide-core experiments, the magnitude of the quenching effect and the details of burst behavior can be calculated directly from basic considerations.

The increase in excursion energy yield that occurred in the Spert I oxide-core tests with a fuel support system that permitted fuel rod bowing is an example of the influence that design details can have on reactor safety and suggests the possibility of using the effect to advantage.

Important differences and similarities were observed in the results of destructive test series with Spert I plate- and pin-type cores. Melting of 3% of the fuel plates in an excursion resulted in only minor transient pressure generation, whereas melting of 35% in the destructive test resulted in pressures of the order of several thousand $lb/in.^2$ with about 10% of the total energy release being attributable to chemical sources and of the order of not more than 3% appearing in mechanical form. The strong mechanical-disassembly effects did not contribute to the quenching of the power burst. In contrast, the rupture of fuel pins in the low-enrichment oxide core test resulted in strong quenching action that caused a change in the burst shape and limited the energy release. Despite the greater reactivity insertion and energy release, the mechanical effects were essentially negligible.

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Abstract

FAST REACTOR SAFETY RESEARCH

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At the time of the First Geneva Conference there was concern over the fact that there might be phenomena which were not understood or even recognized, in effect, that fast reactor safety was not well in hand. For example, the cause of EBR-I meltdown was not understood and the magnitude of the energy release to be expected from a defined accident was under question. The situation is quite different today; all aspects of the behavior of EBR-I have been predicted and demonstrated. Indeed, where the physical behavior of the reactor can be described, the nuclear behavior of fast reactors of the type of EBR-II and Fermi can be predicted. The effect of definable reactivity insertions such as movement of control rods, insertion of hydrogenous material, or a "cold slug" event can be predicted, and can be readily handled by adequate design. However, physical alterations which cannot be well described result in reactivity effects which cannot be well predicted. The major circumstances under which it is difficult to predict physical behavior of material movement and the consequent effect on reactivity is when sodium boiling occurs and there are reactivity changes due to sodium void formation and fuel melting. The flow, gravitational, and other forces under this situation are difficult to predict. The nature of this type of problem, the extent of experimental programs to resolve it, and other associated safety research programs are herein reviewed.

For the abnormal reactor condition in which there is a plugged channel or a loss of heat dissipation capability of the heat transport system, the resulting

sodium temperature increase could cause local sodium boiling, loss of sodium, fuel or fuel clad melting, and consequent fuel movement. The fuel behavior in such incidents has been studied out of pile, using simulant materials, and in-pile, by performing experiments on pins in TREAT. These experiments have lead to a basic understanding of fuel behavior under certain abnormal circumstances. The feasibility of a facility for performing large meltdown experiments has been studied. Such a facility would permit the meltdown of large portions of a reactor core in order to get a more comprehensive picture of sodium void formation, fuel motion, and their effects on reactivity.

In order to determine the theoretical upper limit hazard potential of the fast reactor, a number of studies of the maximum accident have been made. The type of accident analyzed is one in which there is rapid fuel rearrangement under the most severe conditions consequent to a possible but probably not plausible set of circumstances. The most recent analyses take into account the Doppler effect which can be expected. The results obtained indicate that the energy release is substantially less than predicted by earlier analyses.

Although the existing small fast reactors have negative sodium void coefficients, it has been found that large fast reactors of certain designs might have positive sodium temperature coefficients. In order to fully assess the potential of this problem, studies have been made on the stability of fast reactors with positive sodium temperature coefficients and negative structure temperature coefficients. Also, studies have been made on the excursion which might occur upon expulsion of sodium from a reactor with a positive sodium void coefficient and a negative Doppler coefficient. More recently the AEC has supported four studies on 1000MW(e) fast breeder reactor systems in an effort to determine the real degree of difficulty which may result from effects upon reactivity of sodium in such large reactors.

The program of fast reactor safety research in the U. S. in recent years has produced a better understanding of the problems, some refinement of the methods of analysis of the problems, qualitative solutions for some of them, and a much better grasp of the severity of these problems.

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RELEASE AND TRANSPORT OF FISSION PRODUCTS UNDER REACTOR ACCIDENT CONDITIONS

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Data on the release, transport, and removal of fission products released from various fuels under simulated accident conditions are being obtained at over a dozen difference laboratories in the U. S. as a consequence of the general studies sponsored by the USAEC's Reactor Safety Program as well as the more specific investigations undertaken by various reactor programs.

To date most of the work on fission-product release has been directed toward metallic uranium, uranium alloy, and UO_2 fuel materials although limited data are also available on releases from UC and BeO/UO_2 fuels. Data are presented for all fuels as a function of a number of parameters which generally include temperature (up to melting), time (several hours), burnup (up to 11 000 MWd/ton), and ambient atmosphere (steam, air, He, CO_2). Most of the data were obtained out-of-pile using induction; resistance; or plasma-heating techniques although some in-pile results are included.

In general, quantitative releases of the noble gases and iodine may be expected when UO_2 is melted, frequently with significant (up to 100%) release of Cs and Te. Comparable releases are obtained at somewhat lower temperatures ~2000F (1090C) but require several hours. On the other hand, Sr released from UO_2 was $\ll$ 1% except at high temperatures >2900F (1600C) in helium. In both the oxide and the metal, burnup accelerated the rate of release. However, the influence of burnup on the fractional releases of fission products from the metal is considered minor when compared to the other variables. With uranium metal, fractional releases similar to those

from UO_2 are obtained. In addition to release measurements, characterization of the released activity (in terms of its physical and chemical form), its transport (principally deposition), and its removal from air constitute the other principal areas of experimental work. However, the many deposition results reported lend themselves only to a few qualitative generalizations - for example, the nonvolatiles plateout close to the point of release at temperatures above 1830F (1000C).

The characterization studies have emphasized iodine behavior because of its relative hazard potential and have produced interesting, if not completely definitive, results. In helium at 2200F (1200C) iodine is released from the irradiated metal in combined form, which is partly dissociated at higher temperatures and appears as elemental iodine when subsequently oxidized. Particles identified to date ranged from 0.004-μm iodine vapor particles to 250-μm particles resulting from oxidation of uranium in air, these sizes representing the extremes of some not-too-well-defined distributions. In any event, large concentrations of small particle size will soon agglomerate so that the physical characterization is a strongly time-dependent exercise.

Fission-product-removal studies, including sprays, scrubbers, scavenging, and filtration, have been quite extensive and have been quite definitive as far as filters are concerned. Reliable performance of charcoal adsorbers for iodine and particulate filters has been consistently demonstrated with removal efficiencies in excess of 99% even in a steam atmosphere.

By and large, the present investigations are continuing and are being supplemented with larger-scale experiments, for example, the Nuclear Safety Pilot Plant (NSPP), Loss of Flow Test (LOFT), and the Containment Systems Experiment (CSE) which are intended to demonstrate the validity and/or limitation of the extrapolations of the laboratory data as applied to the evaluation of the activity behavior in reactor accidents.

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Abstract

AEROSPACE SAFETY OF ISOTOPIC AND REACTOR POWER SOURCES

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This paper describes the aerospace nuclear safety program being carried out for the United States Atomic Energy Commission by the Sandia Corporation. The program calls for an independent assessment of the safety aspects of the application of nuclear power supplies to aerospace vehicles. The safety program includes analytical modeling, research investigations, ground tests, and reentry flight tests, all designed to reduce areas of quantitative uncertainty which may emerge during the review of the safety plan for a specific system.

One area of investigation which is of concern to most aerospace nuclear power supply applications is that of reentry heating and burn-up phenomena. The problem of survival of reentry heating, both on the relatively steep reentry trajectories of the ICBM and the shallower reentry of the manned orbital flights, has received considerable attention over the past few years. On the other hand, the deliberate use of the mechanisms of orbital reentry heating for the burn-up and dispersal of various materials in various configurations has been of limited interest, except to the designers of aerospace nuclear power supplies. The effort undertaken by Sandia Corporation during the past two years has included the investigation of experimental and analytical techniques for predicting reentry heating and burn-up. These investigations have emphasized the need for improved simulation facilities and for additional material properties studies. Under reentry conditions at elevated temperatures, the properties of the materials used in the power supply appear to be one of the major sources of uncertainty in many of the analyses. Steps are being taken to fill in the gaps in knowledge of such properties for materials used in aerospace nuclear power supply systems. An outline of the materials properties investigation program is presented.

Once burn-up or ablation conditions are reached during reentry heating, the nature of droplet formation and final particle size distribution become of interest for the particular materials under study. Progress of investigations

into the mechanisms of ablation droplet formation is discussed. The current state of knowledge regarding the prediction of the transport, dispersion and deposition of particles introduced into the upper atmosphere by reentry burn-up is described.

Assessment of the safety of the aerospace nuclear power supply under all conditions of handling and transport from fabrication through suborbital flight calls for various ground testing capabilities to simulate the conditions to which the power supply might be subjected. The impact shock conditions into various media have been simulated effectively by various rocket sled techniques. The limited quantitative knowledge regarding overpressure levels and temperature-time histories caused by launch pad explosions and fires has led to a program for the instrumentation of several launch sites to collect this kind of information so that more accurate simulation in ground tests can be achieved.

In order to verify the disassembly features incorporated into a particular aerospace nuclear system by the designer, as well as to ascertain current capability for predicting reentry heating, a flight test was conducted in May of 1963 utilizing a NASA Scout vehicle to place a Sandia payload into a shallow reentry flight on the Wallops Island range. Good telemetered data was received until ionization blackout occurred, and optical data was obtained from ground cameras and aircraft. A summary of this flight test and results are given, and also a brief description of a second reentry flight planned for 1964.

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NORTH AMERICAN EXPERIENCE IN THE RELEASE OF LOW-LEVEL WASTE TO RIVERS AND LAKES

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The controlled release of effluents bearing residual amounts of radionuclides became virtually inevitable nearly two decades ago when man began to produce radioactive substances in substantial amounts. In North America the dominant sources have been the large atomic-energy sites at Hanford, Oak Ridge, and Savannah River in the United States, and Chalk River in Canada. The kinds and quantities of radioactive wastes from these sites are quite different, however, because of the basic differences in major facilities.

Releases have ranged from a few curies of fission products to several thousands of curies of neutron-activation products each day. By contrast, the dozen or more power reactors now operating in North America release, at most, a few millicuries each day in liquid effluents.

Attention devoted to routine surveillance at each site has corresponded with the magnitude of the releases and the potential for radiation exposure of people and other life forms. In addition, the diverse nature of the radioactive wastes, coupled with the wide range of physical, chemical, and biological conditions characteristic of the surface waters at the several sites has permitted research that provides a broad knowledge of the consequences of introducing radioactive materials into aquatic environments and of the ecological relationships that prevail.

This paper will cover the considerable maturity gained during the past five years in these studies, emphasizing several factors of fundamental importance to waste management that have come into sharper focus. These include:

1) Substantial advances in estimating radiation dose to the general public from multiple environmental sources.

2) Identification of the particular environmental pathways that may lead to significant radiation exposure from sites for disposal of liquid waste.

3) Accounting for the quantities of radionuclides present in the water, sediments, and biota by material-balance methods.

4) Notable improvements in instrumentation, including whole-body counters.

5) The very real significance of local variables, such as water quality, and the nature of sediments on the fate of radionuclides.

Although a great variety of fission and neutron activation products have entered lakes and streams, only a few of them have thus far warranted close study because of their active potential for human exposure.

For the most part, fission products, including Sr-90, Ru-106, Ce-144, and I-131 have not been transmitted through aquatic food chains in a manner that would make fish flesh a greater source of exposure to man than the drinking of the water in which the fish were grown. A few activation products, notably P-32 and Zn-65, are important exceptions. Tritium may eventually be the nuclide of greatest significance in drinking water near fuel-reprocessing plants.

A gratifying outcome of the studies that have treated radiation exposure from multiple environmental sources is confirmation that the total dose to the public has been well within appropriate limits at all sites. The general tenor of the environmental studies indicates sound and continuous control of the relevant effluents. The studies should provide a methodology for assuring continued reliable control of liquid effluents associated with the expected growth of the nuclear-power industry in North America.

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ATOMIC ENERGY FOR SOCIETY AND THE BALANCE BETWEEN HAZARD AND GAIN

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The health problems created by atomic energy thus far in all of its manifestations, and for the foreseeable future in its projected peacetime applications, are not significantly different, quantitatively or qualitatively, from many other of the environmental health problems facing our society. The way in which the health problems associated with radiation have emerged, and have continued to be presented to the public, distinguishes - and, in the considered view of many responsible observers, overemphasizes - the hazard of the peacetime applications in relation to the substantial benefits which can accrue to society from an orderly development of the atomic-energy industry.

It should be stressed that, with more than twenty years of experience and some millions of man-years exposure to radiation in the atomic energy industry, and with a medical scrutiny as good as or better than that used for identifying injury to other occupational hazards, not a single instance of injurious effect has been observed in individuals whose exposure has not exceeded the permissible limits. Although injury to radiation has occurred, this has been found only in cases of true accident, where the levels of exposure have greatly exceeded the acceptable threshold limits because of accidental circumstances, due to a failure of man or machine in the same sense as an accidental chemical release or explosion. In applying more sophisticated measures of injury than are customarily used in assessing other occupational hazards, such as shortening of life span or of genetic effect, the concern for the radiation hazard has far outstripped that for other equally potentially hazardous agents. Again, it must be emphasized that no such effects have been demonstrated where present accepted threshold limits have not been exceeded. The implication is not that

we should ignore the evidence of hazard for radiation, but rather should place it in proper perspective in relation to other environmental factors affecting health so we do not unduly penalize a development with such great promise of benefit.

In the area of control of occupational exposures to radiation, much more stringent requirements have been adopted than for other equally hazardous industrial agents. For example, to obtain a relatively small thickness gauge for use in industry, one must register the source with the state agency having responsibility for radiation control, demonstrate the safety precautions for all projected uses, and show that qualified personnel are available to supervise the installation and operation. This same industry can purchase and use many tank cars of benzol or carbon tetrachloride or chlorine, or many other chemicals more reactive and more toxic, with the only restriction that, in some states, if ventilating equipment is used, these plans be approved by the state, and that the official factory inspector, if and when he visits the plant, judges the operation to be safe.

Similarly, the community is apt to regard an atomic energy installation with suspicion and apprehension while accepting or ignoring a chemical operation of equal or greater potential hazard. Different standards of acceptance and performance, and the disordered and confused state of the mechanisms by which society develops the judgments and action in balancing environmental hazards with benefits, can result in discriminatory and even capricious controls. These may even ultimately deprive our society of needed and beneficial goods and services. It is essential that we view the benefits and hazards of atomic energy with the perspective of other comparable activities and their environmental hazards. A number of helpful and provocative examples of such an equilibrium between hazard and benefit, and the mechanisms by which they have become established or accepted in our society, will furnish a useful background for continuing consideration of this relationship with respect to atomic energy.

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U. S. EXPERIENCE IN RADIOACTIVE WASTE MANAGEMENT AND RECENT TRENDS IN RESEARCH AND DEVELOPMENT

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Although the nuclear energy industry in the United States is relatively still in its infancy, significant strides have been made during the past six years in the radioactive waste management field to further reduce the volumes and quantities of radioactive material being discharged to our environment. The initial operation of central station nuclear power reactors during this period has resulted in waste quantities amounting to only a few per cent of original design estimates. However, the adoption of more stringent radiation protection guides in recent years has resulted in a vigorous and expanded research and development program in radioactive waste control, directed at further minimizing the release of radioactivity to our environment and at providing engineered practical systems for utilization in an expanding nuclear energy industry.

Operating experience and data from waste management systems at United States atomic energy facilities such as national laboratories, production sites, reactor testing facilities, and nuclear power reactor installations, are summarized. The efficiency of various treatment systems for large-volume low-activity waste from these operations is evaluated. Recent experience on tank storage of smaller-volume, high-activity waste is briefly discussed, and new management techniques for reducing future tank storage requirements are described.

Ground disposal experience and practice in the United States, which is utilized for certain types of solid and liquid waste effluents, are discussed. Industrial interest in providing waste disposal services for large volumes of solid, low-hazard potential waste continues to increase in the United States. Land burial and sea disposal operations during the past six years are discussed and the recent opening of commercial land burial facilities is noted.

The effectiveness of power reactor waste management systems is described and the extent and impact of these operations, in relation to other nuclear energy effluent control operations, are discussed.

The scope and direction of the U. S. waste management research and developments in synthetic ion-exchange, natural mineral exchange materials, foam separation, and hydrofracturing are discussed as potential future methods for improved control of low and intermediate waste. The possibilities of recycling and/or reusing processed waste streams in laboratory or reactor systems are also described.

The inherent restrictions of tank storage for high activity waste, such as potential leakage and necessity of liquid-waste transfer for periods of hundreds of years, has resulted in an extensive "conversion-to-solids" research and development effort in the U. S. Pilot plant results on various calcination and fixation techniques, such as fluidized beds, heated pots, and radiant heated spray colums, are discussed. The addition of glass-forming materials to high activity waste for the purpose of providing a more stable final product is also being investigated. The long-range waste management programs at plutonium production centers, such as Hanford and Savannah River, with their particular site requirements, are briefly described. The effects of fission product removal for beneficial use is discussed as it relates to long-term waste management operations.

The investigation of specific geologic formations, such as salt structures and impermeable crystalline rock for ultimate storage or disposal of calcined solids or certain types of liquid waste, is described. Plans for a proposed field experiment in salt with full-scale radioactivity are discussed.

Brief reference is made of the long-range engineering studies which are being conducted to evaluate the hazards and economics of various methods for handling, storage and/or ultimate disposal of high and low level liquid and solid wastes. Although treatment and disposal costs are substantial, they should not constitute a major obstacle in the development of a nuclear energy industry in the United States.